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SOUTHERN COMPANY
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HL-6009

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Trip of Reactor Feedwater Pump Results in
Low Reactor Water Level and Manual Reactor Scram

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a trip of the reactor feedwater pump which resulted in low reactor water level and manual reactor scram.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

DMC/eb

Enclosure: LER 50-321/2000-011

cc: Southern Nuclear Operating Company
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SNC Document Management (R-Type A02.001)

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NRC FORM 366 (06-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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TITLE (4)
Trip of Reactor Feedwater Pump Results in Low Reactor Water Level and Manual Reactor Scram

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
										05000
09	29	2000	2000	011	00	10	17	2000		DOCKET NUMBER(S) 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	55		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(ix)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	(912) 367-7851

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

On 09/29/2000 at 1357 EDT, Unit 1 was in the Run mode at a power level of approximately 1520 CMWT (55 percent rated thermal power). At that time, the reactor was scrammed manually following a trip of the operating Reactor Feedwater Pump (RFP) and a decrease in water level. Personnel inserted the scram in anticipation of an automatic scram on low water level. Following the scram, level continued to decrease due to void collapse from the rapid reduction in power and the trip of the RFP resulting in closure of the Group 2 and Group 5 Primary Containment Isolation Valves as designed. Level reached a minimum of approximately 40 inches below instrument zero causing automatic initiation of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) systems. Secondary containment automatically isolated and all four trains of the Unit 1 and Unit 2 Standby Gas Treatment systems automatically started on low level. The RCIC and HPCI systems recovered level; an RFP was placed into service to maintain level. Pressure decreased immediately following the manual scram; no Safety/Relief Valves lifted nor were any required to lift. Pressure was controlled automatically by the Main Turbine Bypass Valves.

This event was caused by component failure. During the implementation of a clearance to isolate a condensate pump, air was introduced into the condensate pump suction header through a leaking pump suction isolation valve. The presence of air led to the trip of the operating RFP on low suction pressure. The valve will be repaired or replaced as necessary during the refueling outage that began on 09/29/2000.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 09/29/2000 at 1357 EDT, Unit 1 was in the Run mode at a power level of approximately 1520 CMWT (55 percent rated thermal power). Operations personnel were reducing power to begin a scheduled refueling outage. Only the 1A Reactor Feedwater Pump (RFP, EIIS Code SJ) was in service because only one RFP was needed to maintain reactor vessel water level at this power level. Operations personnel had placed the 1B RFP in standby, that is, operating on minimum flow and not injecting water into the reactor vessel, and had tripped the 1A condensate pump (EIIS Code SD). Per the approved outage schedule, personnel also were implementing equipment clearance 1-00-20511 on the 1A condensate pump. This clearance contained steps to isolate the pump and electrically de-activate its motor in order to allow the performance of scheduled pump and motor preventative maintenance activities during the outage. The clearance was implemented prior to shutdown in order to allow maintenance personnel to begin work on the pump motor. Work on the pump was scheduled to begin after unit shutdown.

The clearance required the condensate pump suction and discharge valves to be closed, the pump motor power supply breaker to be opened and racked out, and vent and drain valves between the suction and discharge valves to be opened. Opening the vent and drain valves were the last steps on the clearance. When personnel performed these steps, they heard air being drawn into the vent and drain lines and heard the condensate booster pumps (EIIS Code SD) begin to cavitate. Personnel closed the vent and drain valves and notified Operations personnel in the Main Control Room (EIIS Code NA).

When air was introduced into the condensate system, condensate and condensate booster pump discharge pressures decreased causing the operating RFP to trip automatically on low suction pressure. The trip of the RFP caused reactor vessel water level to decrease and the runback of both reactor recirculation pumps (EIIS Code AD) to minimum speed. The licensed Shift Supervisor directed personnel to insert a manual reactor scram in anticipation of an automatic scram on low water level. Operations personnel inserted the manual scram at 1357 EDT with reactor water level at approximately 12 inches above instrument zero. Following the manual scram, water level continued to decrease due to void collapse from the rapid reduction in power and the trip of the RFP. This resulted in receipt of Group 2 and Group 5 Primary Containment Isolation System (EIIS Code JM) isolation signals and closure of the Group 2 and Group 5 Primary Containment Isolation Valves (EIIS Code JM) per design.

Water level decreased to a minimum of approximately 40 inches below instrument zero (118.44 inches above the top of the active fuel) resulting in automatic initiation of the Reactor Core Isolation Cooling (RCIC, EIIS Code BN) and High Pressure Coolant Injection (HPCI, EIIS Code BJ) systems per design. Secondary containment automatically isolated and all four trains of the Unit 1 and Unit 2 Standby Gas

Treatment (EIIS Code BH) systems automatically started on low water level per design. The RCIC and HPCI systems recovered level. With water level above instrument zero and continuing to increase to normal, Operations personnel manually secured the HPCI system. Operations personnel then placed the 1B RFP into service to maintain level and manually secured the RCIC system.

Reactor vessel pressure was within its normal range at approximately 1000 psig at the time of the trip of the RFP. Pressure decreased when reactor recirculation pump speed decreased and again immediately following the manual reactor scram. The Main Turbine Bypass Valves (EIIS Code SO) automatically controlled reactor pressure as designed. No Safety/Relief Valves lifted nor were any required to lift to reduce or control reactor vessel pressure.

CAUSE OF EVENT

This event was caused by component failure. During the implementation of a clearance to isolate the 1A condensate pump, air was introduced into the common condensate pump suction header through a leaking condensate pump suction isolation valve. The presence of air in the system led to the trip of the operating RFP on low suction pressure.

The 1A condensate pump suction isolation valve, closed per equipment clearance 1-00-20511, leaked by its seat. The condensate pumps take suction directly from the main condenser (EIIS Code SQ) hotwell; therefore, the pump suction lines are under a vacuum. When the vent and drain valves located between the closed condensate pump suction and discharge isolation valves were opened per the clearance, water trapped between the closed valves was drawn through the leaking isolation valve into the common condensate pump suction header. This occurred because the area of lower pressure existed on the *upstream* side of the isolation valve by virtue of the vacuum in the main condenser. When the water was gone, air was drawn into the vent and drain lines, past the leaking isolation valve, and into the common condensate pump suction header. The air caused the condensate booster pumps to cavitate and the condensate and condensate booster pump discharge pressures to decrease, reducing the suction pressure available to the operating RFP and causing it to trip on low suction pressure.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. The Reactor Protection System (EIIS Code JC) was actuated manually in response to decreasing reactor water level. The Group 2 and Group 5 Primary Containment Isolation System, an Engineered Safety Feature system, actuated automatically on low reactor water level. The low reactor water level also caused automatic initiations of the RCIC, HPCI, and Unit 1 and Unit 2 Standby Gas Treatment systems and automatic isolation of the secondary containment.

Low reactor vessel water level indicates the capability to cool the fuel may be threatened. Should water level decrease too far, fuel damage could result. Therefore, a reactor scram is initiated to reduce substantially the heat generated in the fuel from fission. The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the emergency core cooling systems, ensures that the peak fuel cladding temperature remains below the limits of 10 CFR 50.46.

The HPCI system is an emergency core cooling system designed to operate in conjunction with the reactor protection system. It is provided to ensure that the reactor is adequately cooled to limit fuel cladding temperature in the event of a small break in the nuclear system and a loss of coolant that does not result in rapid depressurization of the reactor vessel. The HPCI system permits the plant to be shut down while maintaining sufficient reactor vessel water inventory until the reactor vessel is depressurized. The HPCI system continues to operate until reactor vessel pressure is below the pressure at which other emergency core cooling systems can maintain cooling.

To provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barriers, the Primary Containment Isolation System initiates automatic isolation of lines which penetrate the primary containment whenever monitored variables exceed operational limits. A low water level in the reactor vessel could indicate that either coolant is being lost through a breach in the nuclear system process barrier or the normal supply of reactor feedwater has been lost and that the core is in danger of becoming overheated. Low reactor vessel water level initiates closure of various primary containment isolation valves. The closure of these valves is intended to isolate a line breach, conserve reactor coolant, and prevent the escape of radioactive materials from the primary containment.

In this event, the reactor was scrammed manually in response to decreasing reactor water level caused by a trip of the operating RFP on low suction pressure. The Group 2 and Group 5 Primary Containment Isolation System actuated automatically on low reactor water level and the appropriate Primary Containment Isolation Valves closed as designed. The low reactor water level also caused automatic initiations of the RCIC, HPCI, and Unit 1 and Unit 2 Standby Gas Treatment systems and automatic isolation of the secondary containment at their respective setpoints as water level continued to decrease due to void collapse from the rapid decrease in reactor power and the trip of the RFP.

All systems functioned as expected and per their design given the water level transient. The low water level condition was not the result of a line breach or escape of reactor coolant and there was no release of radioactive materials into either the primary or secondary containments. Water level was maintained well above the top of the active fuel throughout the transient and was restored to normal within one minute of the reactor scram. Therefore, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

The 1A condensate pump suction valve will be inspected and repaired or replaced as necessary during the refueling outage that began on 09/29/2000.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

ADDITIONAL INFORMATION

1. Other Systems Affected: No systems other than those mentioned in this report were affected by this event.

2. Failed Components Information:

   Master Parts List Number: 1N21-F010A
   Manufacturer: Henry Pratt Company
   Model Number: XR-70
   Type: Valve, Isolation
   Manufacturer Code: P340

   EIIS System Code: SD
   Reportable to EPIX: Yes
   Root Cause Code: X
   EIIS Component Code: ISV

3. Commitments: No permanent commitments are created as a result of this report.

4. Previous Similar Events: Previous similar events in the last two years in which the reactor was scrammed automatically or manually from power were reported in the following Licensee Event Reports:

   50-321/2000-004, dated 08/04/2000,
   50-321/2000-002, dated 02/25/2000,
   50-321/1999-003, dated 06/01/1999,
   50-366/1999-007, dated 07/27/1999,
   50-366/1999-006, dated 07/14/1999,
   50-366/1999-005, dated 05/27/1999.

   The circumstances surrounding this event, that is, a leaking isolation valve with vacuum on its upstream side combining to draw air into the operating condensate system, are unique to this event. Corrective actions for the previous events therefore could not have prevented this event because they could not have addressed these particular factors.